FENOC

FirstEnergy Nuclear Operating Company

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January 31, 2003
PY-CEI/NRR-2689L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 2002-001-01

Ladies and Gentlemen:

Enclosed is Licensee Event Report (LER) 2002-001 Supplement, "Unplanned Automatic SCRAM During Main Turbine Overspeed Testing". This revision is submitted to correct a minor wording error in the Abstract for clarification purposes. As a result, no reporting criterion was affected by this supplement.

There are no regulatory commitments in this letter. Any actions discussed within this LER are described for information only and are not regulatory commitments. If you have questions or require additional information, please contact me at (440) 280-5294.

Very truly yours,

Vernon K. Higaki

Vernon K. Higaki
Manager - Regulatory Affairs

Attachment

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE Unplanned Automatic SCRAM During Main Turbine Mechanical Trip Weekly Testing

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
09	22	2002	2002	-001	-01	01	31	03	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 90

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

20 2201(b)	20 2203(a)(3)(ii)	50 73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
20 2201(d)	20 2203(a)(4)	50 73(a)(2)(iii)	50.73(a)(2)(x)
20 2203(a)(1)	50 36(c)(1)(i)(A)	✔ 50 73(a)(2)(iv)(A)	73 71(a)(4)
20 2203(a)(2)(i)	50 36(c)(1)(ii)(A)	50 73(a)(2)(v)(A)	73 71(a)(5)
20 2203(a)(2)(ii)	50 36(c)(2)	50 73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
20 2203(a)(2)(iii)	50 46(a)(3)(ii)	50 73(a)(2)(v)(C)	
20 2203(a)(2)(iv)	50 73(a)(2)(i)(A)	50 73(a)(2)(v)(D)	
20 2203(a)(2)(v)	50 73(a)(2)(i)(B)	50 73(a)(2)(vii)	
20 2203(a)(2)(vi)	50 73(a)(2)(i)(C)	50 73(a)(2)(viii)(A)	
20 2203(a)(3)(i)	50 73(a)(2)(ii)(A)	50 73(a)(2)(viii)(B)	

12 LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Kenneth Russell, Compliance Engineer	(440) 280-5580

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	TA	CPLG	GE	Y	B	AA	CPLG	ITT	Y

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	✔ NO				

16. ABSTRACT (Limit to 1400 spaces, i e , approximately 15 single-spaced typewritten lines)

On September 22, 2002, at 0146 hours, when the Perry Nuclear Power Plant was operating in Mode 1 at 90 percent reactor power, a turbine trip occurred which resulted in a turbine control valve fast closure reactor scram Investigation determined that the turbine trip occurred due to the failure of the turbine trip latch assembly to properly reset following weekly testing. While attempting to reset the scram following the event, it was identified that a Scram Discharge Volume (SDV) drain valve stem coupling had failed. The valve repositioned to its required closed position during the scram, however, it would not reopen when the scram was reset Additionally, increased seal leakage from Reactor Recirculation Pump A was noted. The turbine trip was caused by a malfunction of the turbine trip latch assembly which failed to reset following testing of the mechanical trip device. The failure to relatch was determined to be caused by latch travel limitations as a result of two missing setscrews. The missing set screws were replaced, the linkage was adjusted and the assembly was retested. The SDV drain valve failure to open was due to failure of the stem coupling All four SDV vent and drain valve stem couplings were replaced with a coupling of an improved design The reactor recirculation pump seal leakage was due to a dislodged guide bushing in the seal assembly. The seal was replaced with a rebuilt spare seal. This event was reviewed and determined to be within design evaluation limits, and therefore was determined to be not safety significant. This event was reported via the Emergency Notification System in accordance with 10CFR50 72(b)(2)(iv)(B), as an event that resulted in an actuation of the Reactor Protection System when the reactor is critical. This report is being submitted in accordance with 10CFR50.73(a)(2)(iv), any event or condition that resulted in manual or automatic actuation of the system specified in 10CFR50.73(a)(2)(iv)(B)(1), which is the Reactor Protection System.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. Introduction

The Perry Nuclear Power Plant is a Boiling Water Reactor with a General Electric turbine [TA]. The turbine has several trips that are provided to protect the turbine from damage. These trips include mechanical overspeed, electrical backup overspeed and various electrical trips. When a turbine trip occurs, it causes the turbine stop and control valves [TA-SCV] to close. These protective trips are tested by the procedure "Turbine Overspeed Protection Device Trip and Turbine Lube Oil Pump Starts Weekly Test" to ensure they will function properly. This test utilizes a lockout feature to prevent an actual trip of the turbine when the protective feature is tested. A turbine trip causes a reactor scram when reactor power is greater than 38 percent.

On September 22, 2002, at 0146 hours, while performing scheduled activities at 90 percent reactor power, the main turbine tripped and the reactor scrammed while performing weekly testing of turbine protective features. During the turbine mechanical trip test, the turbine trip latch assembly [TA-CPLG] failed to properly reset. This resulted in the depressurization of the turbine Emergency Trip System [TG] header. Pressure switches located in the Emergency Trip System lines for each control valve provide inputs to the Reactor Protection System Each pressure switch provides an input to one RPS channel. The logic is arranged so that actuation of at least one pressure switch in each Reactor Protection System trip system will cause a reactor scram. Failure of the trip latch assembly to properly reset caused a fast closure of the turbine control valves and a turbine control valve fast closure reactor scram.

The Nuclear Regulatory Commission was notified via the Emergency Notification System at 0319 hours on September 22, 2002, (ENF No. 39207), in accordance with 10CFR50 72(b)(2)(iv)(B), as an event that resulted in an actuation of the Reactor Protection System when the reactor is critical. This event is being reported in accordance with 10CFR50.73(a)(2)(iv), any event or condition that resulted in manual or automatic actuation of the system specified in 10CFR50.73(a)(2)(iv)(B)(1), which is the Reactor Protection System.

II. Event Description

On September 22, 2002, at 0146 hours, the plant was operating in Mode 1 at 90 percent reactor power to perform Control Rod System [AA] testing, Reactor Pressure Vessel [RPV] pressure was approximately 1013 psig and normal safety systems were operable except for the Division 2 Emergency Diesel Generator which was in parallel to the grid for testing due to an emergent replacement of the diesel generator governor. During the performance of the procedure, "Turbine Overspeed Protection Device Trip and Turbine Lube Oil Pump Starts Weekly Test," following a mechanical trip test, the turbine trip latch assembly [TA-CPLG] failed to properly reset The turbine trip signal resulted in a turbine control valve fast closure reactor scram

As a result of the scram, RPV level immediately decreased to less than the low RPV level (LEVEL 3, 177 inches) then quickly (about 12 seconds) recovered above the Level 3 setpoint. In order to prevent the high RPV level trip (Level 8, 219 inches) of all feedpumps, proceduralized operator actions were initiated to stabilize RPV level by removing Turbine Driven Feedpumps A and B [SJ] from service. When the first feedpump was manually tripped, the Motor Feedpump automatically started as designed, the second turbine feedpump was then manually tripped. RPV level subsequently decreased to a minimum level of 159 inches before level was stabilized in the proceduralized control band. During the initial level transient, the Reactor Recirculation Pumps [AD-P] downshifted from fast speed to slow speed and the Residual Heat Removal Heat Exchanger second vent valve [BO-VTV] to the Suppression Pool automatically isolated as designed.

When the reactor scrams, all control rods are hydraulically inserted by the Control Rod Drive Hydraulic System [AA]. The water used to insert the control rods is directed to the Scram Discharge Volume (SDV). The discharge volume is equipped with vent and drain valves that automatically isolate during a scram to prevent draining the RPV to the Suppression Pool via the discharge volume. While attempting to reset the scram, it was identified that the SDV was not draining. Subsequently, investigation determined that one of the SDV drain valves would not open due to a failed coupling between the valve stem and its air operated actuator. Separation of the valve from its actuator did not prevent the valve from failing to its safety position, which is closed. The cause of the coupling failure was determined to be an inadequate or incomplete design. This coupling, and three similar couplings on the other drain valve and the two SDV vent valves, were replaced with new couplings of an improved, modified design.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

II. Event Description (continued)

Following the scram, increased leakage was noted on the Drywell Floor Drain Sump level recorder [BD-LR]. The leakage was subsequently identified to be from the Reactor Recirculation System Pump A seal [AD-SEAL]. Although the leakage was still within limits, the seal was replaced to ensure continued reliability for the remainder of the operating cycle.

III. Cause of Event

The scram occurred as a result of the turbine trip. The cause of the turbine trip was due to a malfunction of the turbine trip latch assembly which failed to reset following the testing of the mechanical trip device. The failure to relatch was determined to be caused by missing setscrews in the breakdown spring link assembly that connects the mechanical trip latch assembly to the Emergency Trip System hydraulics. The missing setscrews allowed a fastener on the breakdown spring link nut to limit the stroke of the trip latch rod during reset. Failure of the trip latch rod to fully extend to its reset position prevented necessary latch rod / trip finger engagement such that the latch would not remain reset. The cause of the missing setscrews was not definitively identified.

The coupling for the Scram Discharge Volume valves is a split design with a through bolt that clamps the coupling halves to both the valve and actuator stems. The cause for the failure of the Scram Discharge Volume drain valve to open was a failure of the stem coupling due to inadequate or incomplete design The vendor recommended torque applied to the coupling bolt caused the coupling halves to deflect and cause inadequate stem coupling thread engagement, which resulted in stem coupling failure. The reason the design change was inadequate / incomplete was that incorrect stem coupling material was specified for which the coupling bolt torque was excessive.

The leakage from Reactor Recirculation pump A seal was determined to be due to a guide bushing becoming dislodged from the stationary face holder, preventing the stationary face from applying uniform pressure on the rotating face. Axial pump shaft movement is small but was apparently enough to cause incomplete contact between the 2 seal faces. The as found sealing surfaces indicated that the contact was not uniform along the entire 360 degrees of the seal face (i.e, wider contact pattern in one area and thinner in another). This indicates that the seal holder had become jammed on the side with the dislodged guide bushing thereby preventing the application of equal seal face pressure, resulting in leakage.

IV. Safety Analysis

The Updated Safety Analysis Report (USAR) Section 15 2 3 transient, "TURBINE TRIP" bounds this scram event The USAR analyzed event commences at 105 percent of Nuclear Boiler rated steam flow and normal operating pressure and results in maximum peak power of 114.5 percent power and a maximum peak dome pressure of 1188 psig, which exceeds the conditions that occurred during this event. No safety-relief valves lifted as a result of this event; therefore, the radiological consequences were less than analyzed. During the transient, RPV water level decreased to a low of 159 inches by Wide Range (WR) level indication. All systems responded as described in the bounding analysis. In summary, this event was reviewed and determined to be within design evaluation limits, and therefore was determined to be not safety significant.

A Conditional Core Damage Probability (CCDP) was calculated for the reactor scram due to the turbine trip. All significant systems and components were available except the Division 2 Diesel Generator, which was supplying Division 2 electrical bus, EH12. Two CCDP values were calculated for the event. The first case assumed the diesel was unavailable for 15 days, half the diesel surveillance interval. The calculated CCDP was 8.71E-9. The second case assumed that electrical bus EH12 was lost during the event due to the failure of the inoperable Division 2 Diesel Generator. The calculated CCDP for the second case, using a duration of 1 hour was 3.09E-11. Using NRC guidance of < 1E-6 as a threshold, the event was considered not risk significant.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

V. Corrective Actions

The missing set screws for the turbine trip latch assembly were replaced. The vender did not recommend any additional means of securing the setscrews. Additional components in the assembly were replaced as necessary due to wear. The linkage was adjusted and tested to ensure proper tripping and resetting. Periodic inspections of the linkage are being incorporated into maintenance procedures.

Couplings of an improved design were fabricated and installed in all four SDV vent and drain valves and torqued to the appropriate value.

The seal for Reactor Recirculation pump A was removed and replaced with a reconditioned spare.

The above events and corrective actions have been entered in the Plant Corrective Action Program.

VI. Previous Similar Events

The most recent turbine trip induced scram, LER 98-002, resulted from a failed optical isolation card that caused an initiation of the Reactor Core Isolation Cooling (RCIC) System. The RCIC initiation caused trips of the main turbine and the feedwater turbines. The cause of this event is not similar to this event which was the result of missing fasteners that are part of the turbine-reset mechanism. Therefore, this event is not a recurring event

Energy Industry Identification System (EIIS) codes are identified in the text in the format [xx].